NEXTera ENERGY DUANE ARNOLD

December 7, 2009

NG-09-0907
10 CFR 50.73

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

<u>Licensee Event Report #2009-004-00</u>

Please find attached the subject report submitted in accordance with 10 CFR 50.73. This letter makes no new commitments or changes to any existing commitments.

Christopher R Costanzo

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cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
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NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 8/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. *Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet* e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Duane Arnold Energy Center	05000 331	1 OF 4

4. TITLE

Unplanned Automatic Reactor Scram due to an Invalid Reactor Protection Signal

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER 05000
10	08	09	2009	004	0	12	07	09	FACILITY NAME	DOCUMENT NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (*Check all that apply*)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100%	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	VOLUNTARY LER

12. LICENSEE CONTACT FOR THIS LER	
NAME	TELEPHONE NUMBER (*Include Area Code*)
Bob Murrell, Engineering Analyst	319-851-7900

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT										
CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (*If yes, complete 15. EXPECTED SUBMISSION DATE*) ☒ NO				

ABSTRACT (*Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines*)

On October 8, 2009, while operating at 100% power, during the performance of Surveillance Test Procedure (STP) 3.3.3.2-09B, Reactor Water Level and Pressure Instruments (Loop B) Calibration, an automatic reactor shutdown occurred. The shutdown was caused by the failure to close an instrument isolation valve for a Reactor Vessel Pressure Transmitter. The failure to close this valve resulted in creating a sensed low reactor water level on Reactor Protection System (RPS) channels A2 and B2, and thus resulted in the automatic reactor shutdown.

Primary Containment Isolation System (PCIS) Groups 2, 3 and 4 were received due to reactor water level dropping below 170 inches due to the expected level shrink immediately following the reactor scram. All isolations went to completion. Reactor water level was subsequently returned to normal as expected under post scram conditions.

The cause of this event was that defenses in depth were inadequate to prevent the plant transient during the performance of STP 3.3.3.2-09B.

There were no actual safety consequences and no effect on public health and safety as a result of this event.

I. Description of Event:

On October 8, 2009, while operating at 100% power, the following Limiting Conditions for Operations were in effect:

1. TS 3.7.5, Required Action A.1, 30 day completion time for B Control Building Chiller
2. TS 3.7.2, Required Action A.1, 7 day completion time for B River Water Supply Loop
3. TS 3.3.3.1, Function 6, Required Action A.1, 30 day completion time for position indication of MO-1949B, B Residual Heat Removal Heat Exchanger Vent
4. TS 3.7.1, Required Action A.1, 30 day completion time for 1P22D, D Residual Heat Removal Service Water Pump.

At 1014, Surveillance Test Procedure (STP) 3.3.3.2-09B, Reactor Water Level and Pressure Instruments (Loop B) Calibration, was commenced. Following the performance of a pre-job briefing, Instrument and Control (IC) technicians proceeded to the instruments to be calibrated. As the technicians progressed through the STP they eventually reached STP step 7.1.62 where they were directed to close Reactor Vessel Pressure Transmitter (PT) Isolation valve, PT4564-V-92. Following the completion of Step 7.1.62, the technicians proceeded to and completed step 7.1.63 which required removal of a test fitting tubing cap, connection of test equipment, and the opening of PT4564-V-91, PT4564 Drain/Test Isolation Valve. Upon completion of opening V-91, the technicians noted that the test equipment hand pump hose suddenly pressurized. At 1420 on October 8, 2009, an automatic reactor shutdown occurred. When the technicians heard the scram solenoids de-energize and observed the hand pump tube jump, one of the technicians returned to PT4564-V-92 and manipulated it an additional 1/8 turn in the closed direction.

II. Assessment of Safety Consequences:

There were no structures, systems, or components (SSCs) that failed during this event. Primary Containment Isolation System (PCIS) Groups 2, 3 and 4 were received due to reactor water level dropping below 170 inches due to the expected level shrink immediately following the reactor scram. All isolations went to completion. Reactor water level was subsequently returned to normal as expected under post scram conditions.

This event did not result in a Safety System Functional Failure.

Therefore, the reactor scram did not result in any radiological or nuclear concern which would impact the health and safety of the public.

III. Cause of Event:

A Root Cause Evaluation (RCE 1086) was completed for this event. The RCE determined that the shutdown was caused by the failure to close an instrument isolation valve for a Reactor Vessel Pressure Transmitter PT4564. The failure to close this valve resulted in creating a sensed low reactor water level on Reactor Protection System (RPS) channels A2 and B2, and thus resulted in the automatic reactor shutdown. The specific root cause and contributing factors are as follows:

Root Cause:

Defenses in depth were inadequate to prevent the plant transient, when the IC Technician did not completely close PT4564 V-92 during the performance of Step 7.1.62 of STP 3.3.3.2-09B.

Contributing Factors:

The requirements of the DAEC Technical Procedure Writing Standards related to Critical Steps were not implemented in STP 3.3.3.2-09B.

At the time of issuance of INPO 06-003, Human Performance Reference Manual, administrative procedures for implementing the site's Operating Experience (OE) Program did not require formal review of INPO guideline documents. The document was reviewed outside the OE Program, and actions were implemented to meet the intent of INPO 06-003. The implementation of the critical steps missed the specific area of removing the instrument from service, and focused primarily on the issues identified in OE on returning the instrument to service.

IV. Corrective Actions:

Immediate Actions Taken:

Immediately following the reactor shutdown the site's Maintenance Department initiated a Human Performance investigation, collected written statements from the IC technicians involved with the STP and held an IC Department stand-down to share information regarding the event and to reinforce in-plant job performance expectations.

The plant shutdown was documented in corrective action document CAP 70334, and from this corrective action document Condition Evaluation (CE) 7739, Corrective Action (CA) 53475, and Root Cause Evaluation (RCE) 1086 were initiated.

CE 7739 was written to conduct an initial evaluation under the corrective action program to document the probable cause, immediate, and interim corrective actions for reactor shutdown.

CA 53475 was written to require, as an interim corrective action, IC Maintenance, prior to commencing any STPs, to conduct a review to identify any STPs that include instrument calibrations with shared reference legs. Any that are identified will have the steps that isolate instruments marked as requiring additional verification.

In addition, plant management has determined that all upcoming surveillance testing that is performed on reactor vessel level shared instrument legs (piping) will be regarded as 'high risk' until such time as the corrective actions for RCE 1086 have been completed.

Corrective Actions to Address Root Cause and Contributing Factors:

1. Revise the STPs identified in CE 7778 to require pre-pressurization of the instrument test lines.
2. Revise STP 3.3.3.2-09B to identify critical steps as defined in Technical Procedure Writing Standards.

V. Additional Information:

Previous Similar Occurrences:

A review of LERs over the previous 5 years revealed the following similar occurrences:

LER 2009-001 - Manual Reactor Scram Due to Loss of Condenser Cooling
LER 2009-003 - Unplanned Manual Scram Due to Increasing Reactor Water Level
LER 2007-007 - Reactor Scram Due to 1A2 Non-essential Bus Lockout
LER 2007-005 – Automatic Reactor Scram Due to Scram Discharge Volume High Water Level During Performance of a Surveillance Test
LER 2006-005 - Reactor Scram During Main Turbine Testing

EIIS System and Component Codes:

Primary Containment Isolation System--JM
Engineered Safety Features Actuation System--JE
Level Transmitter--LT
Level Indicating Switch--LIS

Reporting Requirements:

This event is reportable under 10 CFR 50.73(a)(2)(iv)(A).